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6710-97-2276
July 21, 1997

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: Three Mile Island Nuclear Station, Unit I (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
LER 97-007-00

This letter transmits in its entirety, Licensee Event Report (LER) number 97-007-00 which describes the loss of offsite power event that occurred at TMI-1 on June 21, 1997.

This LER is being submitted pursuant to 10 CFR 50.73, using the required NRC forms (attached). NRC form 366 contains an abstract which provides a brief description of the event. For a complete understanding of the event, refer to the text of the report provided on Form 366A.

This event did not adversely affect the health and safety of the public. For additional information regarding this LER contact William Heysek of the TMI Licensing and Regulatory Affairs Department at (717) 948-8191.

Sincerely,

James W. Langenbach

James W. Langenbach
Vice President and Director, TMI

WGH

Attachments

cc: Administrator, Region I- H. Miller
TMI Resident Inspector- S. Hansell
TMI-1 Senior Project Manager- B. Buckley
File 97073

IE22 1

CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

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LANGENBACH,J.W. General Public Utilities Corp.
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SUBJECT: Forwards LER 97-007-00 re loss of offsite power event that occurred at TMI-1 on 970621.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 1+13
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NRC FORM 366 (4-95)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98
1104	LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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TITLE (4)

GENERATOR OUTPUT BREAKER FAILURE RESULTING IN A LOSS OF OFFSITE POWER AND REACTOR TRIP

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
06	21	97	97 --	007 --	00	07	21	97	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	100 %		20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36©(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36©(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
W. G. HEYSEK, TMI LICENSING ENGINEER	717-948-8191

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
B	EA	BKR	1248	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE).	NO				

ASTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On June 21, 1997 at approximately 1214, while the plant was operating at 100% power, TMI-1 experienced a loss of off site power (LOOP) due to the failure of both generator output breakers located in the site's 230KV substation. The fault detected on the #4 bus (230KV) resulted in the opening of the parallel breaker which suffered a re-strike and caused a fault on the alternate #8 bus. The two faults resulted in breaker action which isolated power to the station and resulted in the LOOP. The LOOP resulted in immediate reactor and turbine trips. Both emergency diesel generators started and loaded on to their respective safeguards buses as designed. Without balance of plant power, the condensate, feed, circulating water and main condenser vacuum pumps were not operable. The once through steam generators were fed through the Emergency Feedwater System via two electric pumps powered by the engineered safeguards buses and a steam driven pump. Heat was removed via the steam generator atmospheric dump valves. Loss of station power also deenergized the reactor coolant pumps which forced the plant into a natural circulation mode. Natural circulation flow was achieved within 19 minutes of the trip. Offsite power was restored within 90 minutes. Systems were restored to enable cooling via the main condenser and subsequent restart of the reactor coolant pumps. The failed breakers were replaced with new ABB model 242 PMG 3000 amp breakers. The plant was back on line at 0202 on June 29, 1997.

The root cause of the plant trip and LOOP was the failure of generator breaker GB-1-02 and the re-strike which occurred during the opening of the parallel breaker caused a fault on the alternate #8 bus in the plant 230KV switch-yard. The fault in the "B" phase of GB-1-02 breaker caused severe overheating and ejection of the bushing and conductor from the breaker.

The event is being reported per 10 CFR 50.73(a)(2)(iv)

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

Summary

At 1214 EST on June 21, 1997 Three Mile Island Unit 1 was operating at 100% power generating approximately 830 megawatts. The plant had been on line continuously for nearly 617 days when output breaker GB1-02 [EA/BKR]* failed[1]. The "B" phase of the breaker developed a fault causing severe overheating with subsequent ejection of the bushing and conductor from the breaker housing. This resulted in a fault being detected on the #4 (230 KV) bus. The parallel generator breaker, GB1-12, opened on the fault as required and suffered a re-strike. The re-strike damaged the "B" phase of breaker GB1-12 resulting in a fault on the alternate 230 KV line (#8 bus). Automatic breaker action due to the faults isolated power to the station resulting in a Loss Of Offsite Power. The loss of offsite power resulted in an immediate reactor and turbine trip. Both emergency diesel generators [EK/DG] started and loaded onto their respective safeguards bus as designed. Without balance of plant power the condensate [KA] and feed system [SJ] (including circulating water and main condenser vacuum pumps) was not functional. Once Through Steam Generator (OTSG) [AB/SG] feeding was accomplished through the Emergency Feed system. Heat removal was accomplished through the steam generator atmospheric dump valves. Loss of station power also deenergized the reactor coolant pumps forcing the plant into a natural circulation mode. Verified natural circulation flow was achieved within 19 minutes of the trip. An Unusual Event was declared and the Emergency Response Organization was activated.

Offsite power was restored 90 minutes after the breaker failure. The condensate and circulating water systems were started and main condenser vacuum was restored to enable the return of cooling to the main steam turbine bypass valves [SB/VTV] (dump steam to main condenser) versus the main steam atmospheric dump valves [SB/VTV] (dump steam to the atmosphere). The steam driven emergency feed pump was secured to minimize the possible release paths to the environment. After the main condenser was restored as the heat sink the reactor coolant pumps were restarted, reestablishing forced circulation mode.

The root cause of the GB1-02 breaker failure is thought to be high contact resistance. The root cause of the failure of GB1-12 is thought to be failure to interrupt the arc during opening. These root causes are preliminary and investigation is planned. Generator breakers were replaced with new ABB model 242 PMG 3000 amp breakers. This reduces the probability of this event re-occurring at TMI. The reactor trip was reported in accordance with 10 CFR 50.72(b)(2)(ii).

I. PLANT CONDITIONS BEFORE THE EVENT

Reactor power was at 100% and steady state with the turbine generator producing 830 megawatts and +200 megavars (flowing out). As a result of main transformer heating the var output of the generator was reduced at about 1200 hours (indicated 150 megavars on dispatchers output indication).

II. STATUS OF STRUCTURES, COMPONENTS, OR SYSTEMS THAT WERE INOPERABLE AT THE START OF THE EVENT

There were no structures, components, or systems inoperable at the start of the event that contributed to the event.

III. EVENT DESCRIPTION

Sequence of Events

Event sources as follows:

TMS	Transient Monitoring System
PPC	Plant Process Computer Alarm Printout
Beta	Main Annunciator Panel Alarm Processor[2]
Logs	Operator and STA Interviews and Logs

[1] ITE Model 230 GA 20-30 3000 Amp Dual Pressure SF_6 Gas Breaker

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TEXT (*If more space is required, use additional copies of NRC Form 366A*) (17)

Time	Source	Description
09:56:02.533	PPC	B Main Transformer [EA/XCT] top oil high temperature alarm
09:56:58.533	PPC	A Main Transformer [EA/XCT] top oil high temperature alarm
10:22	LOGS	A & B Main Transformers cooling spray initiated
11:09	LOGS	A & B Main Transformers cooling Spray secured
11:45	LOGS	Raised generator [EA/TG] VARS output to +220 megavars per dispatcher request.
12:00	LOGS	Reduced generator VARS to +200 megavars output per SS direction due to main transformer temperature rise.
12:14:35.379	PPC	Generator Loss of Field Trip
12:14:35.393	PPC	GB1-02 trip (failed generator breaker)
12:14:35.397	PPC	Main Transformer Differential Trip
12:14:35.406	PPC	4 Bus Differential Trip
12:14:35.418	PPC	GB1-12 trip (parallel generator breaker)
12:14:35.464	PPC	8 Bus Differential Trip
12:14:35.562	PPC	Rod Control [AA/JX] Secondary DC Hold Power Failure
12:14:35.595	PPC	Rod Control Main DC Hold Power
12:14:35.634	PPC	Turbine [TA/TRB] Trip
12:14:35.720	PPC	Turbine trip signal trips Reactor
12:14:35.766	PPC	RPS CRD [JC/ZI] Trip Confirm
12:14:35.769	PPC	Reactor [AB/RCT] Trip Isolation
12:14:35.777	PPC	Reactor Trip Confirm
12:14:36.004	Beta	OTSG A EFW [SJ/P] Actuated (Signal only)
12:14:36.008	Beta	OTSG B EFW Actuated (Signal only)
12:14:36.416	PPC	4 or 8 Bus Low Voltage (4 Bus selected)
12:14:36.600	PPC	MU-P-1B [BJ/P] Off, RB Emergency cooling fans [BK/FAN] tripped,7 KV buses deenergized, all Reactor Coolant Pumps [AB/P] tripped, A, B, & C 4KV buses deenergized,
12:14:37.070	PPC	D 4 KV ES Bus undervoltage trip
12:14:37.076	PPC	E 4 KV ES Bus undervoltage trip
12:14:38.150	PPC	T 480 bus undervoltage, NR & NS pumps trip
12:14:39.216	PPC	B OTSG safety [SB/RV] lifts, P, R, & S 480 volt buses trip
12:14:40.366	PPC	D,C,J,K,F 480 volt buss undervoltage trip, A OTSG safeties lift, HDPs [SN/P], COP1s [SD/P], COP2s, & CWPs [KE/P] trip
12:14:41.283	PPC	E,G,H,N,L,F,K,M,B 480 volt buses trip
12:14:42.479	PPC	Loss of Condensate/Booster Pump [SD/P] , MFW pump [SJ/P] trip signal
12:14:42.997	PPC	FW-P-1B Tripped
12:14:44.716	PPC	EG-Y-1B [EK/DG] output breaker closes on E 4KV bus
12:14:44.792	PPC	EG-Y-1A output breaker closes on D 4KV bus
12:14:44:808	PPC	E 4KV ES Bus undervoltage Clear
12:14:44:880	PPC	E 4KV ES Bus ubdervoltage Clear
12:14:47.166	PPC	MU-P-1A running
12:14:47.166	PPC	FW-V-5B [SJ/ISV] closed (MFW isolation)
12:14:48.333	PPC	FW-V-5A closed (MFW isolation)

[2] Beta time stamp was found to be 0.51 seconds ahead of the PPC time stamp. The beta is time synchronozed periodically to minimize this gap. The times reported for the beta in this report have been corrected to PPC time, i.e., beta-0.51 seconds.

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

12:14:50.400	PPC	EF-P-2A [SJ/P] Running (Emergency Feed)
12:14:49.429	PPC	FW-P-1A Tripped
12:14:51.333	PPC	EF-P-2B Running (Emergency Feed)
12:14:51.333	PPC	B OTSG Safeties reseat
12:14:52	LOGS	EF-P-1 Running (Emergency Feed)
12:14:56.881	PPC	Manual Reactor Trip
12:15:03.016	PPC	A OTSG Safeties reseat
12:15:52.450	PPC	MU-V-14B Opened by Primary CRO
12:15:54.416	PPC	MU-P-1C Started Manually
12:26	LOGS	Unusual Event Declared, SS is ED
12:33	LOGS	Natural Circulation Verified
12:46:45.450	PPC	SBO Diesel [EK/DG] Running
12:48:25.466	PPC	C4KV powered by the SBO Diesel [EK/DG]
12:50	LOGS	Risk Counties Notified
12:52	LOGS	Message left for NRC Resident Inspector
12:53	LOGS	NRC (Doug Weaver) contacted via ENS line
12:53	LOGS	NRC Site Resident notified
13:10	LOGS	Dir O&M assumes ED position, Plant Ops Dir assigned as Eplan representative to Substation Yard
13:31	LOGS	ERDS Connected to NRC
13:37	LOGS	4 bus power restored
13:44	LOGS	8 bus power restored
13:51 - 13:53	LOGS	Restored A & B 7 KV buses, A,B, & C 4 KV buses, D,H,G,E,F 480 volt buses
14:06	LOGS	NRC Resident Contacted CR
14:46	LOGS	D 4 KV bus returned to 4 bus
14:46	LOGS	Actual Offsite dose measured near training center is zero net counts per minute for Iodine and Particulate. Scan of plume indicated background.
15:14	LOGS	EF-P-1 secured
15:25	LOGS	NRC Region I Administrator contacted at home by ED to discuss default dose estimate.
15:30	LOGS	E 4 KV bus returned to 8 bus
15:31	LOGS	NRC Resident Inspector briefed in CR by ED
16:14	LOGS	Reactor Trip Isolation bypassed for primary sample.
16:18	LOGS	A Auxiliary Boiler [SA/BLR] fired
16:23	LOGS	C 4KV bus powered from 4 bus, SBO secured
16:48	LOGS	CW-P-1F started
17:01	LOGS	CW-P-1C started
17:07	LOGS	CW-P-1D started
19:34	LOGS	Gland Steam Exhauster started
19:38	LOGS	VA-P-1A & 1C started
19:45	LOGS	VA-P-2A,2B,&2C started
19:50	LOGS	VA-P-1B started
20:15	LOGS	MS-V-4B closed, MS-3A,B, & C opened
20:19	LOGS	MS-V-4A closed, MS-3D,E, & F opened
21:14:23.515	PPC	RC-P-1C started
21:15	LOGS	MU-P-1B started

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

21:16:11.951	PPC	RC-P-1A started
21:18:32.933	PPC	RC-P-1D started
21:20:12.185	PPC	RC-P-1B started
21:24	LOGS	Unusual Event Terminated

Pre-transient RCS conditions were normal with all plant parameters operating within their normal bands. The loss of the 4 bus followed by the 8 bus 0.058 seconds later initiated two parallel chains of events.

Sequence one began with the loss of AC power to the Control Rod Drive (CRD) system. Power to energize the "A" and "B" auxiliary transformers (plant power supply from the grid) is supplied from the 4 and 8 bus. Interruption of this power source deenergized all plant 7KV and 4KV buses. CRD power is supplied from the G & the L 480-volt motor control centers per the scheme below:

8 Bus	4 Bus
A Aux. Transformer	B Aux. Transformer
B 4 KV Bus	C 4 KV Bus
G 480 Volt Bus	L 480 Volt Bus
CRD Main Power	CRD Aux. Power

Power to the control rod drives was lost. This initiated the reactor shutdown portion of the transient. The reactor trip confirmed signal initiated a turbine trip signal, however the turbine had been tripped by a different scheme.

Sequence two was initiated by the main transformer differential current trip. The transformer trip resulted in a generator trip and a turbine trip. The release of the load on the turbine caused an immediate increase in speed to about 1877 rpm. The turbine master trip bus indicated the turbine tripped 0.0864 seconds before the reactor trip.

The loss of offsite power deenergized the "A" and the "B" 7 KV buses. These buses power the reactor coolant pumps and are equipped with an automatic transfer function if power is lost to either bus. The 4 bus (230 KV) deenergized initiating an auto-transfer. However, since the 8 bus was deenergized 0.058 seconds later, power to the four reactor coolant pumps was lost. Reactor coolant flow began to decrease upon loss of power to the pumps, but due to the design of the pumps a significant coastdown was observed. For both reactor coolant loops, coastdown was effectively complete after two minutes. Signs of coastdown were exhibited in the "A" reactor coolant loop for approximately six minutes and in the "B" loop for approximately eight minutes. This coastdown coupled with high decay heat resulted in a rapid transition to natural circulation.

The transition to natural circulation is defined in ATP 1210-10 (Abnormal Transient Procedure). The guideline for determining whether natural circulation exists looks for five items:

Stable RCS ΔT of approximately 30 to 50°F
Hot leg temperature (Th) < 600°F
Incore Thermocouples tracking Th
Cold Leg Temperatures approach saturation temperature for secondary side pressure, i.e., steam generators are coupled to primary side (reactor coolant)
Verify heat removal from the OTSGs

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

Natural circulation builds in as a function of the heatup of water in the core. As this water is heated it expands and becomes less dense. The higher density water in the loop then exerts a force on the lower density water in the core similar to a manometer. This force develops until a sufficient ΔP (change in pressure) exists to allow the higher density cooler water to push the lower density hotter water out of the top of the core. The head loss experienced in passing through the core and up the hot leg must be overcome by the ΔP created. The coastdown from the reactor coolant pumps aided greatly in maintaining the hot legs warmer than the cold legs while water in the core continued to heat up. High decay heat and the coastdown stimulated a rapid and efficient transition to natural circulation.

The first indication post trip was the collapse of the normal 100% power ΔT (change in temperature across the core) from 45°F to approximately 15°F. The fact that ΔT did not fall to values near zero indicates the coastdown continued until the initiation of natural circulation. The rapid initiation of emergency feed-water provided the immediate heat removal necessary to increase the density of water in the OTSG tubes and eventually in the cold legs. Consequently, at no time during this event did reactor coolant flow decay to zero. The coastdown provided flow while natural circulation built in and natural circulation built in prior to the decay of coastdown flow.

For 70 seconds post trip Th and Tc both decreased. During this time RCS flow was reduced by greater than 90% of its normal value. After 70 seconds Th began to increase and Tc continued to decrease. The continued decrease in Tc was due to the cooling effect of emergency feedwater. The upward direction turn in Th was due to the heatup of water in the core and greatly reduced flow rates. As this warmer water was pushed out of the core Th continued to increase. This trend continued until cooler water from the cold leg (not the steam generator) replaced the warmer water in the core and the hot leg. This cool water entering the core removed sufficient heat to reduce the slope of the incore temperature rise. Th and incore temperatures continued to rise for 5.88 minutes in the "A" loop and 4.6 minutes in the "B" loop. At this time the hot leg temperatures on both loops turned and began decreasing. This indicates that cool water had progressed through the core and up the hot leg. At this time (6 minutes post trip) core ΔT was 44° F in "A" loop and 49° F in the "B" loop and natural circulation existed. Loop ΔT remained essentially constant for the duration of the natural circulation period.

Five and a half minutes from the start of the Th's decrease incores made a significant step down in temperature. This decrease was the first of the emergency feedwater cooled reactor coolant to reach the incores, completing another phase of the first loop turnover. Verified Natural Circulation was called by the Shift Technical Advisor 19 minutes post trip.

Fifteen minutes post trip the "A" cold leg temperature dropped approximately 12° F in a thirty second time frame. This temperature drop did not directly correlate to any operational stimulus. With OTSG pressures unchanged and the decrease in cold leg temperatures the "A" steam generator moved across the saturation line on the PT plot. An OTSG on the subcooled side of the saturation line indicates a heat source to the reactor coolant system rather than a heat sink. Since gland sealing steam comes off the "B" steam generator it was being fed harder and steamed harder. The effect of pulling so much heat from the "B" OTSG explains why the "A" OTSG was not removing much, if any, heat. This condition persisted until the restoration of forced circulation. At this time the "A" OTSG immediately returned to the superheated side of the saturated curve on the PT plot indicating a resumption of normal heat removal.

After natural circulation flow was determined to exist a limited cooldown was established. This cooldown was to insure natural circulation flow would continue to develop without threat of stalling. In addition to this aspect, the LOOP removed power from the pressurizer heaters. With this added limitation operators had to minimize the insurge into the pressurizer. The general operating technique was to adjust cooldown to maintain a stable pressurizer level and a constant reactor coolant pressure. Efforts to accomplish this were successful while cooldown rate averaged less than or equal to 30°F/hour.

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

Reactor Coolant Pumps were restarted after normal heat removal was restored. That is, when the heat sink was the main condenser and not the atmosphere. This required the restoration of condensate, main condenser vacuum, and circulating water. Once these systems were restored the reactor coolant pumps were restarted and the emergency plan condition was terminated.

The transition back to forced circulation was accomplished by starting the "C" reactor coolant pump. The pump restart immediately mixed coolant from the hot leg and the cold legs such that the core ΔT went to zero. This resulted in a ten degree collapse in "B" loop temperatures to a Tave of approximately 505° F. An approximate 20° F increase in "A" loop cold leg temperature with a 10°F decrease in the "A" loop hot leg temperature to a Tave of approximately 505°F. The "B" loop pump was started first because of the lower ΔT in that loop. The "A" loop experienced some reverse flow during the time when only one reactor coolant pump was running. The remaining reactor coolant pumps were started within the next eight minutes. During this time the average coolant temperature increased to 510°F as the "A" loop water was fully mixed with the balance of the reactor coolant.

Reactor coolant pressure dropped 30 psig at the first pump start. This is not an unusual phenomenon since the total volume in the cold legs is greater than that of the hot leg. Therefore the average coolant temperature will not be the average of the hot and cold leg temperatures. It will be a weighted factor toward the cold leg temperature. This effect coupled with newly restored pressurizer spray capabilities accounts for the decrease in pressure.

At the start of the event, main turbine reached an estimated maximum speed of approximately 1877 rpm. A jump in speed from 1800 to some higher value could have been missed by the turbine speed scanning software, however the coastdown from that speed would be indicated in the data. The turbine coastdown was captured in its entirety indicating it began at 1877 rpm. Consequently, the maximum indicated turbine speed achieved during the loss of offsite power event was 1877 rpm.

Main Feedwater response to the loss of off site power was typical for the conditions. Power was lost to the condensate and booster pumps when the 4 KV Balance of Plant (BOP) buses tripped. Consequently, both main feedwater pumps tripped on counter circuit interlocks. A loss of both feedwater pumps signal was then sent to the Reactor Protection System (RPS) cabinets. By this time the reactor was already tripped. Main feedwater isolation (FW-V-5A/B) occurred as expected on the "A" & "B" main feedwater pumps after diesel generators restored power. MFW flow coasted to zero within seconds.

Immediately post trip the OTSG response was normal for the event. OTSG pressure increased to the safety setpoint immediately post trip as designed. The main steam safeties functioned as designed to limit the OTSG pressure and provide for an initial RCS cooldown to normal post trip conditions. The safeties on the "A" OTSG remained open for 22.65 seconds and the safeties on the "B" OTSG remained open for 12.12 seconds. In response to high EFW flow rates OTSG pressures decreased as the cool EFW water quenched the steam in the steam generator, thereby dropping pressure. As EFW flow was decreased OTSG pressure responded by increasing. As EFW flow was manually throttled OTSG pressure was slowly reduced over the next two hours to facilitate natural circulation. This pressure reduction was accomplished by control of the MS-V-4A/B valves (Atmospheric Dump Valves) in conjunction with EFW flows at approximately 100 gpm.

The "A" and "B" OTSG pressure responses were similar, however the PT plot (Pressure Temperature plot) indicated the "A" steam generator may not have contributed to the RCS cooling for the entire event as explained above.

Equipment Problems

Control rod trip insertion times for four rods exceeded the Technical Specification limit of 1.66 seconds See LER 97-008. The cause of the problem is a hydraulically induced effect resulting from reduced clearances in the old style control rod drive thermal barriers due to the presence of deposits on the internal check valves and between the thermal barrier bushing and the leadscrew.

GPU Nuclear will continue current reactor plant chemistry controls and extended rod exercises as well as complete the replacement of all remaining old style thermal barriers with modified thermal barrier design prior to startup from the 13R outage.

The Plant Process Computer (PPC) Pressure Temperature (PT) plot locked up initially upon startup and numerous times thereafter. The PPC was continuing to update the time on the PT plot however, the values on the PT plot and the PT plot elapsed time were not updated. To aid in restoring the PT plot to use the STA canceled the existing copy running and restarted it. Upon restart the PT plot would run for some time period and fail as described. The PT plots were re-created from the data stored by the PPC.

The condenser vacuum pump exhaust high range radiation monitor, RMG-25, failed post trip. Trouble shooting indicated the radiation detector was in working order. Investigation indicates the radiation monitor may have failed due to the hot moist environment of the vacuum exhaust piping that developed after the loss of the vacuum pumps.

During the loss of offsite power event, two RCS pressure instruments [AB/PI] Plant Process Computer (PPC) indication failed to zero. RCS pressure instruments PT-963 (Loop A Wide Range) and PT-949 (Loop B Wide Range) are powered through the signal conditioning cabinets and were not directly affected by the loss of offsite power, however the value sent to the PPC failed to zero. The PPC input from these pressure instruments feeds through the Diverse Scram System (DSS) cabinet and this system was de-energized by the loss of offsite power.

During the loss of offsite power electrical power was lost to all BOP and engineered safeguards buses. The vital buses were then powered from the batteries via the inverters. The main annunciator panel [IB/PL] is controlled by the Beta control unit, which is powered from the "C" and the "D" vital buses. After the engineered safeguards buses were re-powered the vital bus power supply shifted from the batteries [EJ/BTRY] to the battery chargers [EJ/BYC] via the inverters [EJ/INVT]. The ultimate power supply for the vital buses was then the diesel generators. At this time the alarm lights which were lit on the main annunciator panel in the control room began to cycle in intensity. The period of the cycle was approximately 10 seconds from dim to bright and back to dim. This phenomenon has been seen in the past when a power source feeding either vital bus powering the Beta was changed. The source of the slow strobe effect is thought to be a minor frequency phase misalignment. The relative intensity of the cycle varied over the period on the diesels. The slow strobe effect continued until offsite power was restored. The alarms were functional and dod not impact the ability to maintain the plant in a safe and stable post trip condition.

Root Cause of Reactor Trip

TMI-1 generator output enters the grid via two parallel output breakers to two 230 KV buses. Generator output to the 4 bus passes through the GB1-02 breaker and output to the 8 bus passes through the GB1-12 breaker. Generator Breakers GB1-02 and GB1-12 have been in service since 1974 and were rebuilt in 1991. The rebuild was primarily due to SF6 gas leaks. Preventive maintenance was performed on both breakers in 1993 and 1995. Test results met the specifications.

On 6/19/97 a relay technician noted that the current reading on the B phase of GB1-02 was lower than A and C phases and the current on the B phase of GB1-12 was higher than its A and C phase currents. The following were the readings on the demand meters in the Substation Control Building and the primary currents based on a 3000/5 current transformer:

Phase	GB1-02	GB1-12
A	1.7 (1020 amps)	1.97 (1182 amps)
B	0.7 (420 amps)	3.5 (2100 amps)
C	1.8 (1080 amps)	1.9 (1140 amps)

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

The relay technician reported this condition to engineering. Engineering discussed this condition with relay supervision. This unbalanced current indication was not perceived as a significant equipment problem. The accuracy of the current transformers at the low end of their range was questioned. The consensus was to set up a current transformer check and a thermal scan on Monday June 23, 1997.

At 12:14 on June 21, 1997 a phase to ground fault developed on the B phase pole of the GB1-02 circuit breaker (#4 bushing). Indications are that the fault occurred in this location, but the actual reason for the failure cannot be determined due to the extent of breaker damage. High contact resistance is known to occur on breakers and prior to the failure this breaker was carrying less than half of the full load current on the center or "B" phase.

The fault was detected by the #4 Bus Primary and Backup Differential, TMI-1 230 KV Residual Lead Differential, and TMI-1 Unit Differential schemes. The operation of these schemes took place between 0.42 and 1.32 cycles following the inception of the fault and resulted in tripping GB1-02, GB1-12, 105102, 109112, 1B12, TMI-1 Turbine, and TMI-1 Field Breaker. The digital fault recorder traces for the TMI-1 19.9 KV B and C phase current traces and the TMI-1 230 KV Neutral current show that the field circuit breaker position operated 3.8 cycles into the event. The excitation decay in the main generator continued to provide current flow to the fault. The Digital Fault Recorder (DFR) indicates that this fault current continued to flow for 17.7 seconds from the start of the event. This is expected for the TMI-1 design.

As GB1-12 was interrupting the fault it is believed that a re-strike or failure to interrupt occurred, creating a B phase to ground fault within GB1-12. This theory is supported by data from the digital fault recorder, especially the voltage traces for the 230 KV lines feeding the TMI 230 KV station. As GB1-12 is interrupting the fault the voltage on the 1091, 1092, and 1051 lines and the #8 Bus voltage starts to be restored as would be expected following a successful interruption of the fault. The voltage was quickly suppressed to zero indicating either a re-strike in GB1-12 and/or the inception of a subsequent fault. It is believed that the B phase pole of GB1-12 flashed to ground at this point. Test results indicated a failure of the B phase and internal inspection confirmed that an internal fault had occurred on GB1-12. The #8 Bus Primary and Backup Bus Differential Schemes operated since the fault in GB1-12 was within their zone of protection. The operation of these schemes resulted in tripping of breakers 109102, 109202, and 1B02. Operation of the backup bus differential relay scheme was detected by the digital fault recorder.

After 12.4 cycles from the start of the fault, the GB1-02 Breaker Failure Scheme operated. This operation cannot be explained by the location of the fault or the continued source to the fault from TMI-1 generator. Subsequent testing showed that the breaker failure scheme timer worked properly.

Description of Breaker Failure

Substation Bus 4 and Bus 8 tripped on differential. The Unit Differential, Main Transformer Differential and Main Transformer Ground Overcurrent relays operated. Approximately 10 minutes after the trip, B phase entrance bushing and associated conductor was ejected from breaker GB1-02.

The generator fed the fault for approximately 17 seconds after the generator trip. B phase generator current increased from about 26.5 Kamp to 51.87 Kamp. C phase current on the generator also increased because of the delta to wye connection of the main transformer.

GB1-02 Damage: The number 4 bushing was found ejected from the breaker. The upper porcelain was, for the most part, intact, lying on the ground next to the breaker. The center conductor, with leads still attached, was hanging from the steel structure of a disconnect switch about 45 feet from the breaker. The wire was wrapped around the steel structure. Pieces of porcelain were found farther away from the breaker. A large burn spot was observed on the tank for B phase. The burn spot is in the vicinity of what would normally be the bottom of the bushing. The paint is discolored for a section about 24 inches high and about 48 inches around the edge of the B phase tank. There was extensive damage to the contact assembly on the generator side of the breaker. GB1-12 Damage: There is no physical damage evident from the outside of the breaker, however test results were unsatisfactory. Internal inspection revealed that a fault had occurred. The SF6 arc by-products are a personnel safety concern that limits root cause evaluation at this time. The breaker will be cleaned by the vendor and will be inspected for any further indication of root cause.

The rating of the GB1-12 breaker was compared to the conditions at the time of the fault. The breaker is rated to interrupt 43 Kamps. Opening under fault condition is a severe stress on the breaker.

Summary

The root cause for the fault in GB1-02 is concluded to be high contact resistance leading to an internal fault in the breaker. The root cause for the fault in GB1-12 is failure to interrupt the arc or restoration of the arc (re-strike) after the breaker opened.

IV. COMPONENT FAILURE DATA

Breaker Failures- ITE Model 230 GA 20-30 3000 Amp Dual Pressure SF6 Gas Breaker, Short Circuit Capability 43 Kamps,
Control Rod Drives- detailed discussion on the slow rod drop times is provided in LER 97-008,
Safety Parameter Display System PT Plot- The SPDS PT plot at TMI is a site specific software package,
RMG 25 failure- Victoreen Model 846-2
RCS Pressure Instruments- RC-PT-963 and RC-PT-949 are Rosemount INC / Emmerson pressure transmitters.
Model number 1153GD9PA
Main Annunciator Panel Brightness Oscillations- Hathaway Systems Corporation, Beta Products Division, Model Betalog 4100

V. AUTOMATIC OR MANUALLY INITIATED SAFETY SYSTEM RESPONSE

During this transient three safety system elements actuated. EFW auto start on loss of RCPs (Reactor Coolant Pumps) or loss of Main Feedwater Pumps, HSPS isolation on low OTSG pressure, and emergency diesel generators started on loss of power. The EFW auto start occurred as designed and is described below. The emergency diesel generators auto-started on an undervoltage signal. The "D" 4 KV bus was without power for a total of 9.310 seconds and the "E" 4KV bus was without power for a total of 9.232 seconds. The emergency diesel generators repowered the ES buses in less than 10 seconds as designed. HSPS isolated the FW-V-92B valve on low pressure (less than 600 psi) as designed, after operators lowered OTSG pressure during the initial cooldown (approximately two hours post trip) without bypassing the HSPS feedwater isolation signal. FW-V-92A was closed manually, shortly thereafter. This had no effect since main feedwater was secured and steam generator cooling was being accomplished through the emergency feedwater system.

EFW flow in previous trips has been documented as having very little modulated control. EFW is typically on full flow or off EFW flow behaved in very much the same manner for this event. However, EFW was in hand for most of the event and therefore did not show its reactionary control to the same extent.

VI. ASSESSMENT OF THE SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

Plant Systems

All safety systems actuated as designed and within allowable time tolerances. The Reactor Protection System functioned correctly, providing a reactor trip signal. However, the control rods had alreadt safely shut down the reactor when power to the control rod drive motors was interrupted. LER 97-008 discusses the safety implications of the control rods that failed to meet technical specifications minimum flight time. The EFW (emergency feedwater) system and the emergency diesel generators functioned as designed fulfilling their respective safety function.

This event was the first unplanned transition to natural circulation in the history of TMI-1. The transition to natural circulation was prompt and smooth.

Operational

The post trip and safety system response was as expected. Operator response to this event was appropriate. Operators commenced a cooldown of approximately 30°F per hour to insure that natural circulation would not be disrupted. A cooldown required boration to cold shutdown concentrations. This was accomplished and the cooldown was continued to approximately 505°F at which time it was terminated.

Radiological

A "Non-Routine Effluent Release" occurred from the release of steam from the main steam safeties, the operation of EF-P-1, and the use of the Atmospheric Dump Valves (MS-V4A/B) to remove heat from the primary plant. The first dose projections that were obtained used RM-G-26 and RM-G-27 readings. These instruments typically read between 20 and 50 counts per minute as a result of background radiation and their check sources. Using the normal instrument readings and an eight hour release duration, the RAC (Radiological Assessment Coordinator) calculated an offsite dose of about nine mRem Committed Dose Equivalent (CDE) child thyroid dose and about 0.2 mRem total effective dose equivalent. These projections were appropriate as a bounding calculation since there could have been an undetectable increase in primary to secondary leakage. Later, a contingency calculation was performed assuming that the RCS activity and the primary to secondary leakrates remained as they were prior to the event[3]. In this case, the calculated doses were 1.5 E-5 mRem CDE. Field team readings indicated zero mR/hr exposure outside the plant.

The total curie release to the environment from the event was 0.024 curies primarily composed of tritium. The maximum offsite dose calculated from this release was less than 0.005 percent of the ODCM (Offsite Dose Calculation Manual) quarterly limit. The extremely tight (low leakage) condition of the steam generators and the complete lack of fuel defects played a major role in minimizing the total radiological release to the environment during this event.

Emergency Plan

The ED chose to activate the entire IREO to provide the additional support that was necessary to recover from the event. Response from the IREO was such that all of the facilities (ECC, TSC, & OSC) were manned within the required time.

[3] After plant restart and return to 100% power primary to secondary leakrates were comparable to pre-event values (very low).

VII. PREVIOUS EVENTS OF A SIMILAR NATURE

There have been no previous Loss of Offsite Power events at TMI-1.

VIII. CORRECTIVE ACTIONS

The cause of the event was the failure of the 230 KV generator output breakers. The root cause of the breaker failures is not absolutely known, however these breakers will be decontaminated of arc byproducts and studied. Corrective actions either completed or planned are outlined below:

Completed actions:

1) Both failed breakers have been replaced by new breakers of a different manufacturer.

2) Control rod drive slow insertion times actions are outlined in LER 97-008.

3) The control room global silence alarm function was repaired and is now restored to normal use.

4) Current balance on the generator breakers and gas pressures in the remaining two ITE circuit breakers is being checked daily. These checks will continue until investigation of the breaker failures is complete.

Planned actions:

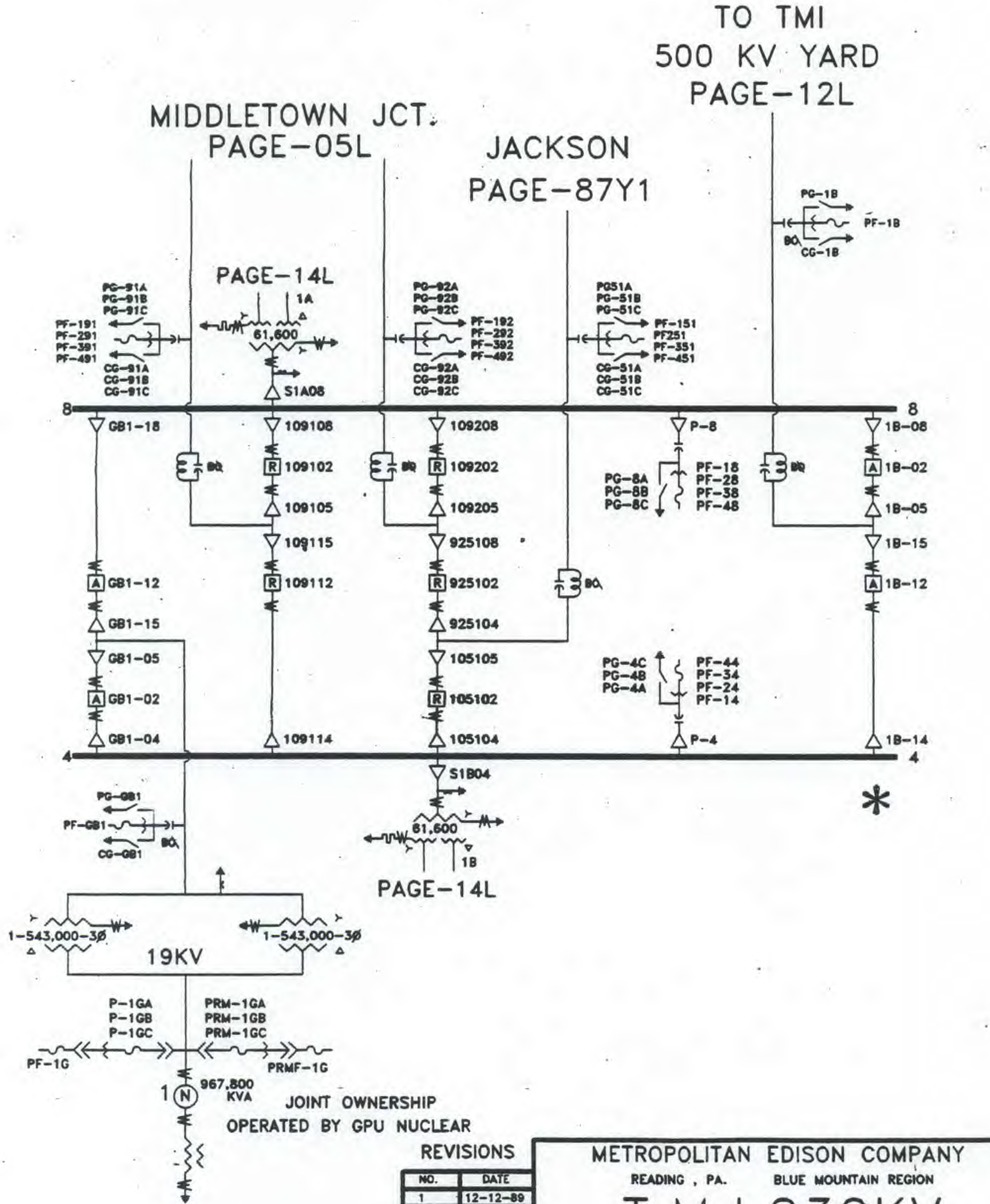
1) The Computer Applications department is currently investigating the PT plot failure to update. This action is occurring under Corrective Action Process (CAP) 1997-0401.

2) Action associated with the failure of two RCS pressure instruments output data to reach the PPC (Plant Process Computer) is occurring under CAP 1997-0395.

3) The Main Annunciator Panel (Beta) brightness oscillation effect is under investigation through CAP form 1997-0460.

4) RMG-25 failure to function in adverse conditions is being investigated under CAP form 1997-0403.

* The Energy Industry Identification System (EIIS), System Identification (SI) and Component Function Identification (CFI) Codes are included in brackets, "[SI/CFI]", where applicable, as required by 10 CFR 50.73(b)(2)(ii)(F).



REVISIONS

NO.	DATE
1	12-12-89
2	11-29-94
3	09-25-95

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